

REACTOR PHYSICS ASSESSMENT OF THE INCLUSION OF UNSEPARATED NEPTUNIUM IN MOX REACTOR FUEL

Ronald J. Ellis

Oak Ridge National Laboratory
Oak Ridge, Tennessee, USA
ellisrj@ornl.gov

Keywords: reactor physics, mixed-oxide fuel, MOX, neptunium, Np, SCALE, TRITON

ABSTRACT

Reducing the number of actinide separation streams in a spent fuel recovery process would reduce the cost and complexity of the process, and lower the quantity and numbers of solvents needed. It is more difficult and costly to separate Np and re-combine it with Am-Cm prior to co-conversion than to simply co-strip it with the U-Pu-Np. Inclusion of the Np in mixed oxide (MOX) fuel for light water reactor (LWR) applications should not seriously affect the operating behavior of the reactor, nor should it pose insurmountable fuel design issues. In this work, the U, Pu, and Np from typical discharged and cooled PWR spent nuclear fuel are assumed to be used together in the preparation of MOX fuel for use in a pressurized water reactor (PWR). The reactor grade Pu isotopic vector is used in the model and the relative mass ratio of the Pu and Np content (Np/Pu mass is 0.061) from the cooled spent fuel is maintained but the overall Pu-Np MOX wt% is adjusted with respect to the U content (assumed to be at 0.25 wt% ^{235}U enrichment) to offset reactivity and cycle length effects. The SCALE 5.1 scientific package (especially modules TRITON, NEWT, ORIGEN-S, ORIGEN-ARP) was used for the calculations presented in this paper. A typical Westinghouse 17x17 fuel assembly design was modeled at nominal PWR operating conditions. It was seen that U-Pu-Np MOX fuel with NpO_2 and PuO_2 representing 11.5wt% of the total MOX fuel would be similar to standard MOX fuel in which PuO_2 is 9wt% of the fuel. The reactivity, isotopic composition, and neutron and γ sources, and the decay heat details for the discharged MOX fuel are presented and discussed in this paper.

1. INTRODUCTION

Reducing the number of actinide separation streams in a spent fuel recovery process would reduce the cost and complexity of the process, and lower the quantity and number of solvents needed. It is more difficult and costly to separate Np and re-combine with Am-Cm prior to co-conversion than to simply co-strip it with the U-Pu-Np.¹⁻³ Inclusion of the Np in MOX fuel for LWR applications should not seriously affect the operating behavior of the reactor, nor should it pose insurmountable fuel design issues. A reasonable U-Pu-Np mixed-oxide (MOX) composition is seen to be suitable as a means of utilizing unseparated Np as part of MOX utilization in a pressurized water

reactor (PWR), which would otherwise require expensive Np separation. If Np can be kept with the Pu, this would represent an important improvement.

The SCALE 5.1 code system⁴ is used for the calculations for this paper. For this paper, the first step in the study was to model a typical light water reactor (LWR) fuel composition used in a standard PWR reactor for a representative operating history using the TRITON depletion sequence⁵⁻⁷ in the SCALE 5.1 computer code system. TRITON models are used to simulate the power history and the cooling after discharge of the nuclear fuel and to provide a representative spent fuel composition which serves as a source for the MOX fuel considered in this paper. A U-Pu MOX mixture with a PuO₂ mass fraction of 9 wt% was the reference MOX case, and various U-Pu-Np MOX compositions were then assessed in PWR simulations.

In this work, the isotopic composition of low enriched uranium (LEU) (4 wt% U-235) PWR fuel burned to 50 GWd/MTIHM (metric tonne of initial heavy metal), followed by 10 years of cooling, was taken to be typical discharged LWR nuclear fuel, and the U, Pu, and Np from this material were considered to be used in the production of MOX. These actinide elements were kept in their relative elemental ratios as in the cooled LEU PWR discharged fuel, and assumed to be used together in newly-prepared MOX fuel, for utilization in a PWR. A typical Westinghouse 17×17 fuel assembly design and PWR operating conditions were modeled. Various (U, Pu, Np) MOX compositions, characterized by various MOX wt% loadings, were assessed in PWR simulations. The relative ratio amounts of Pu and Np are constant, but the overall amount of Pu+Np MOX is adjusted with respect to the U content to offset reactivity and cycle length effects. A reasonable U-Pu-Np MOX composition is seen to be suitable as a means of using the recovered Np as part of the MOX irradiation, which would otherwise require expensive separation.

Np-237 is a very mobile actinide; therefore reducing its presence in spent nuclear fuel (SNF) waste is desirable for waste repository considerations. Also, burning unseparated Np-237 in MOX fuel, in addition to the advantages to the SNF recycling/separation processes, also reduces the inventory of separated Np. Though Np-237 is a fissionable nuclide, its inclusion with Pu in the MOX fuel further degrades the fissile fraction of the reactor-grade Pu MOX, which is a beneficial non-proliferation aspect of this approach.

2. TRITON MODEL FOR ASSESSING MOX FUEL IN A PWR

Different sequences in SCALE 5.1 were used for the assessment of MOX compositions in PWR reactors. The TRITON depletion sequence was used in simulating the fuel power history and depletion for a Westinghouse 17×17 PWR fuel assembly over three fuel cycles, to a burnup of 60 GWd/MTIHM. The TRITON control module was originally developed in tandem with the NEWT functional module of SCALE to support two-dimensional (2-D) transport and depletion calculations. TRITON can be used to provide automated, problem-dependent cross-section processing followed by calculation of the neutron multiplication factor for a 2-D configuration using NEWT. Additionally, this functionality can be iterated in tandem with ORIGEN-S depletion calculations to predict isotopic concentrations, source terms, and decay heat as a result of time-varying fluxes calculated in a 2-D deterministic fashion or in a 3-D stochastic approach.

The power history spanned a total of 4.25 years, with down time between the three fuel cycles. Taking advantage of assembly symmetry, one quarter of the assembly was modeled with TRITON, as shown in Fig 1. Additional isotopic information, neutron and γ source details, and decay heat contributions (total and by nuclide) were calculated as required with the ORIGEN-ARP sequence, using data generated in the TRITON cases.

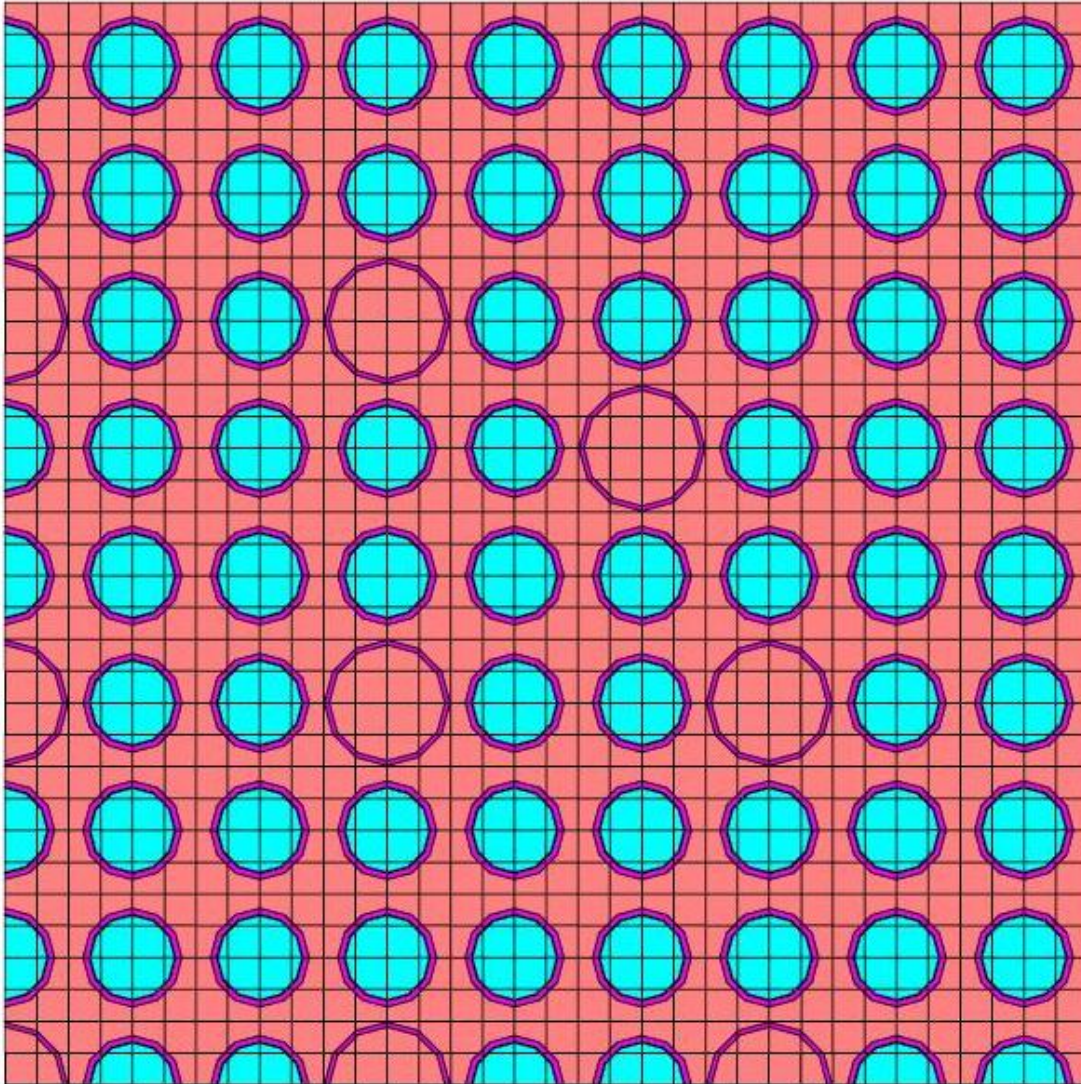


Fig.1 Quarter-view: PWR W17×17 fuel assembly model used with SCALE/TRITON

In the TRITON model, the fuel was represented as an initially uniform distribution of homogenized MOX fuel material in all rods of the standard W17×17 fuel assembly. The assembly includes 264 fuel rods and 25 guide tubes. Fig 1 shows the layout of the rods in the fuel assembly and the positions of the water-filled guide tubes.

3. RESULTS OF ASSESSMENT OF NEPTUNIUM IN MOX FUEL

The series of TRITON results corresponding to various loadings of Pu+Np (oxides) in the MOX fuel were analyzed for reactivity behavior and spent fuel characteristics (decay heat, neutron and γ sources, etc.). Fig 2 shows plots of the infinite medium multiplication constant (k_{∞}) as a function of fuel burnup for a variety of MOX loadings, including the reference 9wt% Pu MOX.

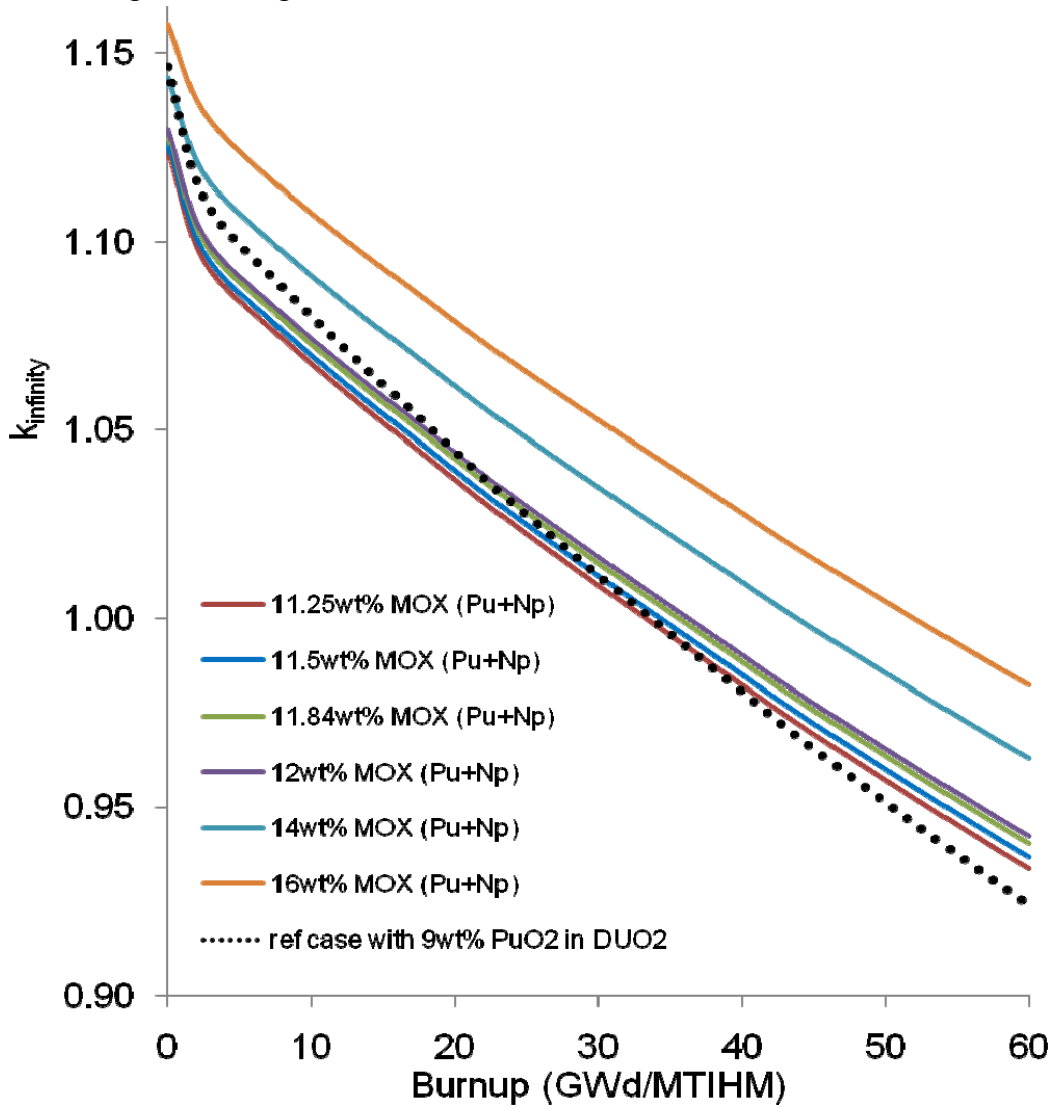


Fig.2 Behavior of PWR reactivity with burnup for the Pu and Pu+Np MOX fuel

As seen in Fig 2, the reference MOX case without initial Np has a higher initial reactivity (in relation to its MOX loading) than the various Pu+Np MOX cases, but k_{∞} declines more quickly in this case. The initial Np in MOX leads to the production of the very fissile Pu-239 during the reactor operation, which offsets the decline in k_{∞} . Np-237 captures a neutron to form Np-238; Np-238 undergoes β decay (half-life of 2.1

days) to Pu-238, which undergoes neutron capture to Pu-239. Np-237 is produced through a sequence of reactions and decays starting with neutron capture by U-235, as well as through neutron capture on U-238 and other reactions, and through Am-241 α decay. The use of depleted uranium helps reduce the production of Np-237.

In this work, the various MOX fuel cycles were compared by integral k_{∞} . These values were determined for each case for a postulated discharge burnup of 60 GWd/MTIHM. The corresponding value of integral k_{∞} for the reference 9wt% Pu MOX case was set as the target value for a desired Pu+Np MOX mixture. It was determined that Pu+Np MOX fuel at 11.5wt% MOX in DUO₂ has the same integral k_{∞} as the reference case. Fig 3 shows the reactivity (k_{∞}) versus burnup curves for this Pu+Np MOX case compared to the reference Pu MOX case. These curves clearly show that k_{∞} drops off more slowly with burnup for this Pu+Np case than for the reference MOX case, owing to the net build-up of Pu-239.

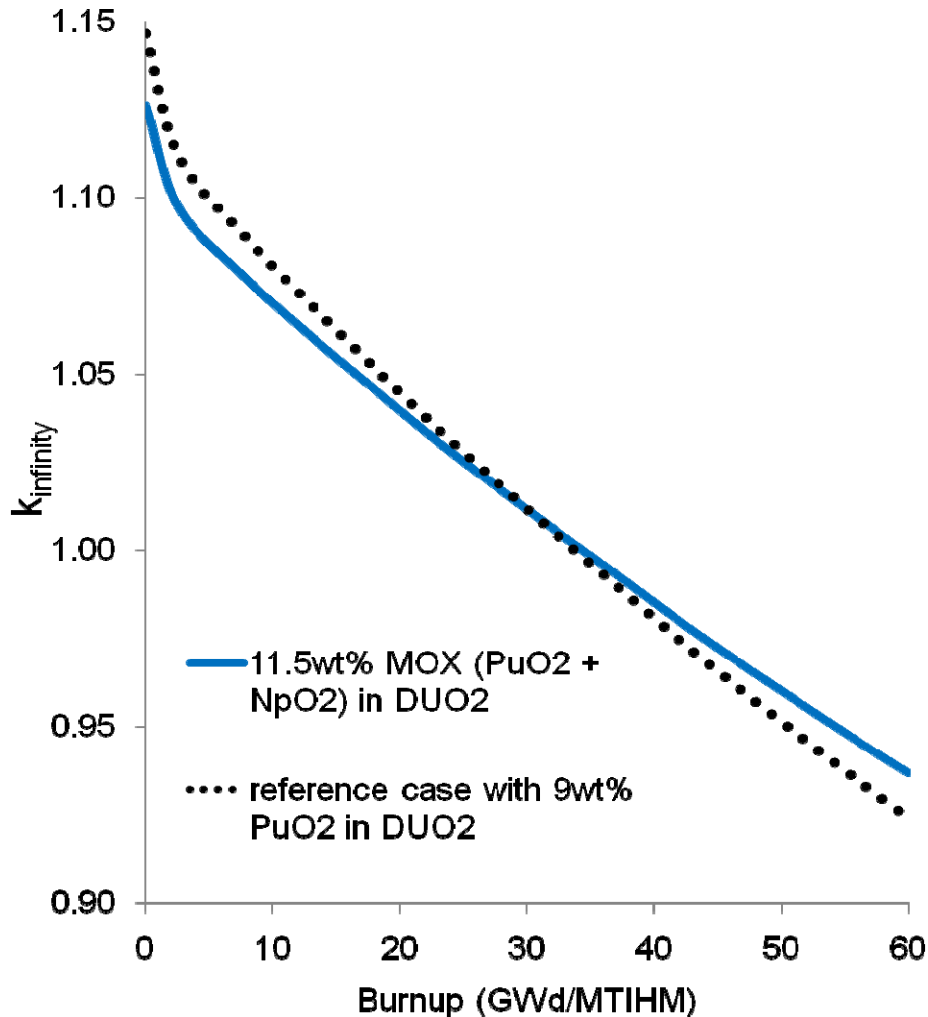


Fig.3 PWR reactivity with burnup for 11.5wt% Pu+Np MOX fuel compared to reference 9wt% Pu MOX fuel

3.1 Isotopic Compositions of Discharged MOX fuel

Table 1 lists the main actinides in the MOX fuel at the beginning of the first fuel cycle and at discharge at a burnup level of 60 GWd/MTIHM for both the reference MOX case and the case with 11.5wt% MOX (PuO₂+NpO₂) in DUO₂. The change in mass content for each nuclide is indicated as well as the elemental masses for U, Np, Pu, Am, and Cm.

Table 1. Isotopics of MOX fuel with and without initial Np (per MTIHM).

Nuclide	11.5wt% MOX (PuO ₂ +NpO ₂) in DUO ₂ (discharge at 60 GWd/MTIHM)			9wt% MOX (PuO ₂) in DUO ₂ (discharge at 60 GWd/MTIHM)		
	Mass in g/MTIHM			Mass in g/MTIHM		
	BOC	Discharge	Mass change	BOC	Discharge	Mass change
u-232	0	9.47E-03	9.47E-03	0	7.30E-04	7.30E-04
u-233	0	6.03E-03	6.03E-03	0	1.09E-03	1.09E-03
u-234	1.52E+01	1.16E+02	1.00E+02	1.57E+01	5.85E+01	4.28E+01
u-235	2.21E+03	1.18E+03	-1.04E+03	2.28E+03	1.07E+03	-1.20E+03
u-236	1.02E+01	2.50E+02	2.40E+02	1.05E+01	2.67E+02	2.56E+02
u-237	0	2.28E+00	2.28E+00	0	2.53E+00	2.53E+00
u-238	8.83E+05	8.46E+05	-3.63E+04	9.07E+05	8.69E+05	-3.84E+04
u-239	0	5.24E-01	5.24E-01	0	5.63E-01	5.63E-01
u elem.	8.85E+05	8.48E+05	-3.70E+04	9.10E+05	8.70E+05	-3.93E+04
np-237	6.61E+03	3.10E+03	-3.52E+03	0	2.21E+02	2.21E+02
np-238	0	5.58E+00	5.58E+00	0	4.57E-01	4.57E-01
np-239	0	7.56E+01	7.56E+01	0	8.12E+01	8.12E+01
np elem.	6.61E+03	3.18E+03	-3.43E+03	0	3.02E+02	3.02E+02
pu-238	3.07E+03	5.39E+03	2.32E+03	2.55E+03	2.11E+03	-4.39E+02
pu-239	5.91E+04	3.16E+04	-2.75E+04	4.91E+04	2.34E+04	-2.56E+04
pu-240	2.71E+04	2.26E+04	-4.50E+03	2.25E+04	1.80E+04	-4.56E+03
pu-241	1.05E+04	1.32E+04	2.66E+03	8.71E+03	1.10E+04	2.29E+03
pu-242	8.65E+03	9.36E+03	7.06E+02	7.19E+03	8.29E+03	1.11E+03
pu-243	0	9.95E-01	9.95E-01	0	1.00E+00	1.00E+00
pu-244	0	3.07E-01	3.07E-01	0	3.07E-01	3.07E-01
pu elem.	1.08E+05	8.21E+04	-2.63E+04	9.01E+04	6.28E+04	-2.72E+04
am-241	0	1.32E+03	1.32E+03	0	1.01E+03	1.01E+03
am-242m	0	3.98E+01	3.98E+01	0	2.97E+01	2.97E+01
am-242	0	1.18E+00	1.18E+00	0	1.09E+00	1.09E+00
am-243	0	2.57E+03	2.57E+03	0	2.38E+03	2.38E+03
am-244	0	1.39E+00	1.39E+00	0	1.44E+00	1.44E+00
am elem.	0	3.93E+03	3.93E+03	0	3.42E+03	3.42E+03
cm-242	0	2.04E+02	2.04E+02	0	1.92E+02	1.92E+02
cm-243	0	9.56E+00	9.56E+00	0	9.09E+00	9.09E+00
cm-244	0	1.54E+03	1.54E+03	0	1.57E+03	1.57E+03
cm-245	0	1.83E+02	1.83E+02	0	1.88E+02	1.88E+02
cm-246	0	1.43E+01	1.43E+01	0	1.78E+01	1.78E+01
cm-247	0	4.03E-01	4.03E-01	0	4.99E-01	4.99E-01
cm elem.	0	1.95E+03	1.95E+03	0	1.98E+03	1.98E+03

The change in mass content for each nuclide is indicated as well as the elemental masses for U, Np, Pu, Am, and Cm.

3.2 Neutron and Gamma Source Strength in Discharged MOX Fuel

The neutron source strength in SNF is taken to be the sum of the (α,n) source and the neutron source due to spontaneous fission (SF), with the latter usually being the much larger component. For typical SNF, the main sources of neutrons at discharge are Cm-244 and Cm-242. At the time of discharge, the SF neutron source for Cm-244 is about four times greater than that for Cm-242. At very long times (more than 10,000 years) after discharge, the Pu-242 SF source dominates. The half-life of Cm-242 is relatively short (approximately 0.45 years) while the half-life of Cm-244 is longer (about 18.1 years). By 5 years after discharge, SF in Cm-244 is the main source of neutrons in the SNF. As shown in Table 1, it is apparent that the 11.5wt% Pu+Np MOX fuel at discharge actually has a smaller Cm content than the reference discharge 9wt% Pu MOX fuel. Table 2 tabulates the neutron source (per MTIHM) for the Pu+Np MOX and the reference Pu MOX cases, at time of discharge (taken to be 0.001 years), and then after 5, 10, and 30 years of cooling. In addition, the relevant information from a representative LEU PWR case is included for comparative purposes.⁸

Table 2. Neutron source for discharged MOX fuel.

	discharge	5 years	10 years	30 years
	n/s/MTIHM	n/s/MTIHM	n/s/MTIHM	n/s/MTIHM
9wt% MOX (reference)	2.28E10	1.51E10	1.25E10	5.92E9
11.5wt% (with Np) MOX	2.24E10	1.46E10	1.21E10	5.75E9
<i>LEU PWR</i> (42.2 Gwd/MTU)	<i>1.06E9</i>	<i>5.10E8</i>	<i>4.23E8</i>	<i>2.05E8</i>

Photons are emitted by many fission products (FP) and some actinide nuclides in SNF. At discharge, the strongest source of γ radiation is Np-239. After 5 years of cooling, the strongest γ sources are Ba-137m and Cs-134. Table 3 lists the γ source (per MTIHM) for the Pu+Np and reference Pu MOX cases, as well as the LEU PWR example results, at discharge, and then after 5, 10, and 30 years of cooling.

Table 3. Gamma source for discharged MOX fuel.

	discharge		5 years		10 years		30 years	
	γ /s/MT	MeV/s/MT	γ /s/MT	MeV/s/MT	γ /s/MT	MeV/s/MT	γ /s/MT	MeV/s/MT
9wt% MOX reference	3.28E18	1.01E18	1.97E16	8.68E15	1.08E16	4.60E15	6.25E15	2.26E15
11.5wt% (with Np) MOX	3.26E18	1.03E18	1.98E16	8.51E15	1.11E16	4.58E15	6.66E15	2.27E15
<i>LEU PWR</i> (42.2 Gwd/MT)	<i>2.94E18</i>	<i>8.56E17</i>	<i>1.32E16</i>	<i>5.51E15</i>	<i>7.68E15</i>	<i>3.05E15</i>	<i>4.29E15</i>	<i>1.58E15</i>

For the neutron source considerations, it is seen that the 11.5wt% Pu+Np MOX fuel has similar, though actually slightly lower neutron source levels than the reference MOX up to at least 30 years after discharge. In terms of the γ source, the 11.5wt% Pu+Np MOX fuel is comparable to the reference MOX case in terms of γ /s/MTIHM, and the corresponding γ energy (MeV/s/MT) is almost identical to the reference MOX case.

3.3. Decay Heat in Discharged MOX Fuel

As mentioned, decay heat of the discharged fuel assemblies is an important consideration. In the ORIGEN calculations of the decay heat, all actinide nuclides and FP nuclides in the discharged fuel assemblies are considered. The decay heat behavior for the 9wt% reference MOX (Pu) fuel and the 11.5wt% Pu+Np MOX fuel are presented in Fig 4. Table 4 presents total decay heat results for certain cooling times, for these two considered cases and, for a comparison, the results for a representative LEU PWR case.

Table 4. Total decay heat (W/MTIHM) for 60 GWd/MTIHM discharged MOX fuel.

	discharge	5 years	10 years	100 years	1,000 years	10,000 years	100,000 years
9wt% Pu MOX	3.15E5	8.12E3	6.40E3	2.23E3	4.74E2	9.38E1	6.71E0
11.5wt% (Pu+Np) MOX	3.21E5	9.97E3	8.25E3	3.37E3	5.80E2	1.19E2	1.09E1
<i>LEU PWR (42.2 GWd/MT)</i>	<i>2.48E5</i>	<i>2.25E3</i>	<i>1.49E3</i>	<i>3.95E2</i>	<i>7.30E1</i>	<i>1.74E1</i>	<i>1.44E0</i>

At discharge, the major decay heat contributors are Np-239, Cm-242, Rh-106, and Cm-244. After 5 years of cooling, the largest decay heat sources are Cm-244, Pu-238, and a number of FP nuclides, particularly Ba-137m and Cs-134. By 30 years after discharge, Am-241 is the largest decay heat source. At 100 years, Pu-240 followed by Am-241 are the main decay heat contributors. At very long cooling times (100,000 years) the main decay heat source is Pu-239, followed by Pu-242, U-234, and Np-237. Polonium isotopes are also small long-lasting decay heat sources.

For the representative LEU PWR case, the main decay heat nuclides at discharge are Np-239, Pr-144, Cm-242, and Cs-134; the discharge burnup of the LEU case is only about two-thirds that of the MOX cases, so the actinide content in the LEU SNF is quite different. By five years, the LEU SNF main decay heat contributors are Pu-238, Cm-244, Am-241, and Cm-242. At 100 years, the major decay heat nuclides are Am-241, Pu-240, and Pu-239. As in the two MOX cases, Pu-239 is the major decay heat nuclide by 100,000 years after discharge.

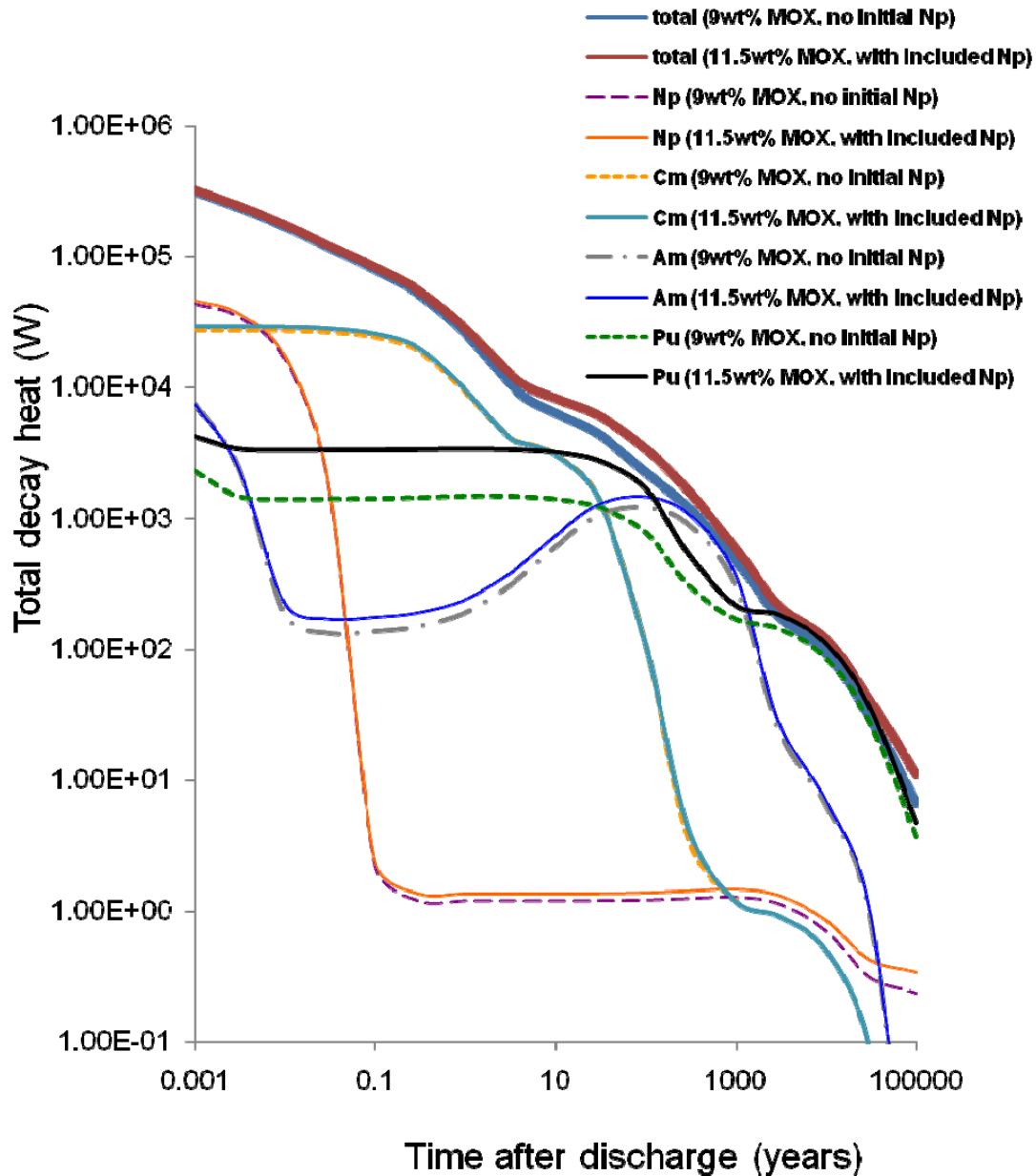


Fig.4 Total and certain component decay heat power for 60 GWd/MTIHM discharged 11.5wt% Pu+Np MOX fuel and 9wt% Pu MOX fuel

In Fig 4, the total decay heat is shown for both MOX cases as are the contributions from the Np, Cm, Am, and Pu isotopes. The decay heat values in this figure are per MTIHM. As is evident from Fig 4, the decay heat for the Pu+Np MOX case is slightly greater than that for the reference Pu MOX case; this difference is partly a result of the included Np but also because the MOX weight fraction is greater for the Pu+Np MOX. The main differences in the actinide decay heat contributions are a long-term increase in Np decay heat for the Pu+Np MOX fuel, a greater mid-term and long-term contributions from Pu, and a greater mid-term contribution from Am.

As seen in Fig 5, the decay heat from the discharged 11.5wt% MOX fuel that contains initial Np is greater than the decay heat from the reference 9wt% Pu MOX. A major contributor to the Pu+Np MOX decay heat is the content of Pu-238 (2.5 times greater than for Pu MOX fuel) in the spent fuel, and 50% more Pu-239 at discharge in the Pu+Np MOX fuel.

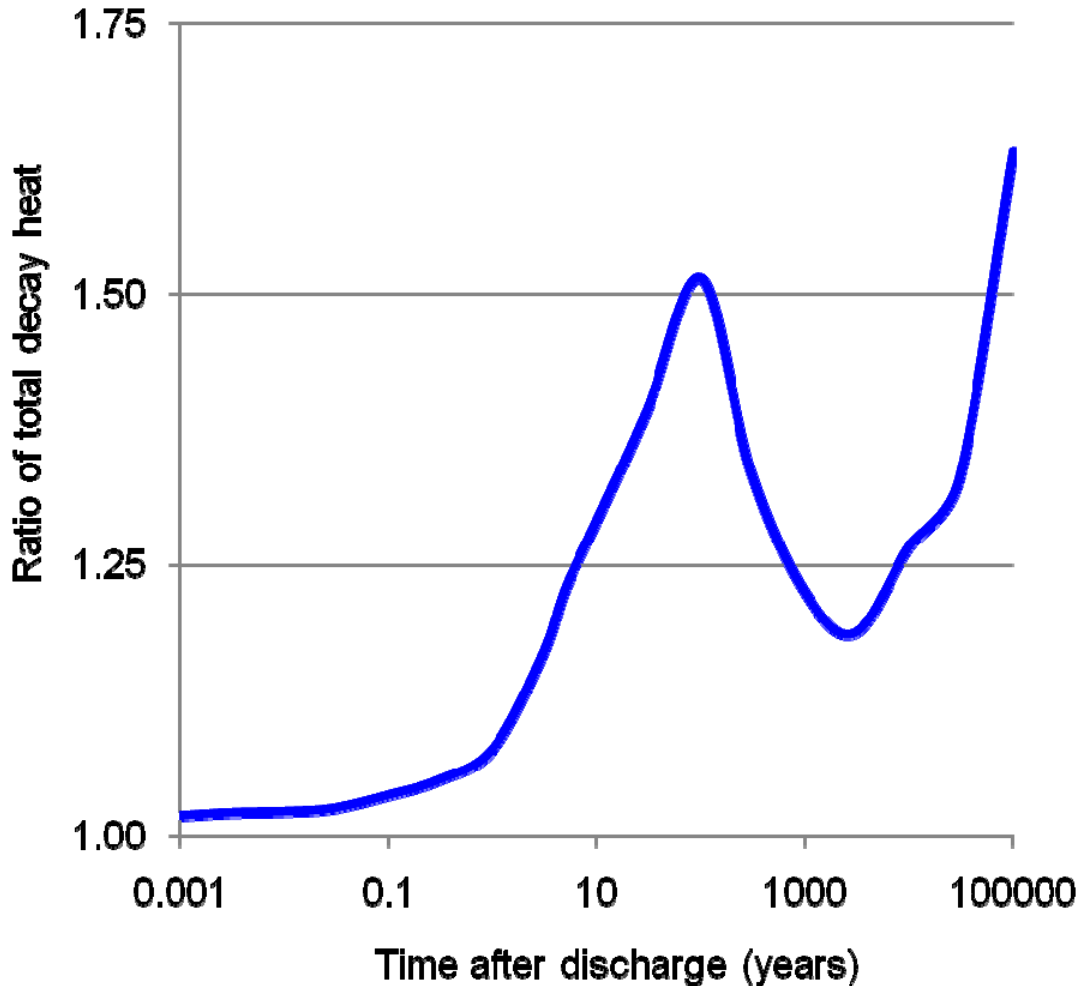


Fig.5 Ratio of total decay heat power from 60 GWd/MTIHM discharged 11.5wt% Pu+Np MOX fuel to discharged 9wt% Pu MOX fuel

3.4. Properties of U-Pu-Np MOX fuel

From a fuel performance standpoint, the physical properties of the U, Pu, and Np oxides are all similar.⁹ The dioxides of Pu and Np have the same structure as UO₂. The addition of Pu in MOX reduces the thermal conductivity by ~8%. Thus, the centerline temperatures will be higher in Pu MOX than in LEU UO₂ and as a result the fission gas release will be slightly higher, but this can be accommodated in fuel rod design (lower fissile content / larger plenum volume). The addition of a small percentage of Np (the Np/Pu mass ratio for this work is only 0.061; Np is 0.66wt% of the heavy metal) will not

significantly affect the fuel performance (which should be confirmed in irradiation tests of U-Pu-Np oxide fuels). Thermal conductivity correlations are generally good to $\pm 10\%$ so the small effects from the addition of NpO_2 will not be significant.

5. SUMMARY AND CONCLUSIONS

In terms of reactivity considerations, the effects of including Np with Pu in MOX fuel can be offset simply by increasing the weight fraction of the MOX material in the fuel. From a fuel performance perspective, a small amount of NpO_2 should have little effect on the predicted performance of the MOX fuel pellets. At Oak Ridge National Laboratory, through the modified direct denitration (MDD) process in the coupled end-to-end (CETE) demonstration,¹⁻³ U/Pu/Np powders were obtained from LEU SNF and test pellets were pressed. U/Pu/Np is made by partial partitioning/decontamination of used LWR fuel and in general will have U-235 content slightly above natural levels.¹⁰ U/Pu/Np test pellets could be subjected to test irradiations in the future, such as in irradiation facilities¹¹ (thermal, and fast) in the HFIR reactor.

In this work, a MOX case based on 9wt% Pu MOX in 0.25wt%-enrichment DUO_2 was considered as a reference. The reactor-grade plutonium isotope vector in this reference MOX fuel corresponds to a PWR fuel with 4wt%-enriched UO_2 burned to 50 GWd/MTHM and cooled for 10 years. Methods have been applied previously to a range of actinide and MOX considerations in a number of reactor types.¹²⁻¹⁷ Based on this methodology, a series of TRITON cases was completed in the current paper for the variation of MOX loadings of Pu+Np MOX in the fuel. The integral k_{∞} was determined for each case for a conjectured scenario with discharge burnup at 60 GWd/MTIHM. It was determined that a Pu+Np MOX at 11.5wt% had the same integral k_{∞} as the reference MOX case.

The neutron and γ source characteristics of the spent Pu+Np MOX fuel and the decay heat behavior of the fuel were compared against the reference MOX case. For the neutron source considerations, it was found that the 11.5wt% Pu+Np MOX fuel has similar, though actually slightly lower, neutron source levels to those of the reference MOX up to at least 30 years after discharge. The γ source for the 11.5wt% Pu+Np MOX fuel was comparable to the reference MOX case in terms of $\gamma/s/MTIHM$, and the corresponding γ energy (MeV/s/MT) was almost identical to the reference MOX case. The decay heat from the 11.5wt% Pu+Np MOX fuel was somewhat greater than the reference MOX decay heat - about 40% greater in the 10 to 1000 year time interval and about 50% greater in the 10,000 to 100,000 year time frame; the long-term difference is mostly due to the greater amount of Pu-238 and Pu-239 present in the fuel at that time.

REFERENCES

1. L.K. Felker, R.J. Vedder, E.A. Walker, E.D. Collins, "Product Conversion: The Link between Separations and Fuel Fabrication", Proceedings of the Atalante 2008 Conference, Montpellier, France, (May 2008).
2. E.A. Walker, R.J. Vedder, G.L. Bell, "Characterization of Mixed-Oxide Powder and Pellets from Modified Direct Denitration", The American Nuclear Society Annual Meeting, June 2008, Anaheim (2008).

3. E.D. Collins, et al., "Closed Nuclear Fuel Cycle Technologies to Meet Near-Term and Transition Period Requirements", Proceedings of the Atalante 2008 Conference, Montpellier, France (May 2008).
4. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations, ORNL/TM-2005/39, Version 5.1, Vols. I-III, Oak Ridge National Laboratory (2006). Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-732.
5. M.D. DeHart, M.D., "TRITON: A Two-Dimensional Depletion Sequence for Characterization of Spent Nuclear Fuel," Sect. T1 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations, ORNL/TM-2005/39, Version 6, Vols. I-III, Oak Ridge National Laboratory (2006).
6. M. D. DeHart and S. M. Bowman, "Improved Radiochemical Assay Analyses Using TRITON Depletion Sequences in SCALE," Proceedings of International Atomic Energy Agency Technical Meeting "Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition," August 29–September 2, 2005, London, United Kingdom. IAEA-TECDOC-CD-1547, Session 2, pp. 99–108 (May 2007).
7. Mark D. DeHart, "High-Fidelity Lattice Physics Capabilities of the SCALE Code System Using TRITON," The American Nuclear Society and the European Nuclear Society 2007 International Conference on Making the Renaissance Real, November 11–15, 2007, Washington, D.C. Trans. Am. Nucl. Soc. 97, 598–600 (2007).
8. B. Hyland, R.J. Ellis, G.R. Dyck, G.I. Maldonado, J.C. Gehin, G.W.R. Edwards, "Transmutation of Americium in Light and Heavy Water Reactors", Paper 9244, To be presented at Global-2009, Paris, France, September 2009.
9. L.J. Ott, Oak Ridge National Laboratory, private communications, December 2008, and March 2009.
10. D.G. Del Cul, Oak Ridge National Laboratory, private communications, March 2009.
11. Ronald J. Ellis, Jess C. Gehin, Joel L. McDuffee, and Randy W. Hobbs, Oak Ridge National Laboratory, "Analysis of a fast spectrum irradiation facility in the high flux isotope reactor," paper 533, *Proc. of PHYSOR'08 International Conference on the Physics of Reactors "Nuclear Power: A Sustainable Resource,"* Interlaken, Switzerland, September 14–19, 2008.
12. T. Greifenkamp, G.I. Maldonado, J.C. Gehin, M.D. DeHart, "Analysis of Minor Actinide Target Depletion in a Pressurized Water Reactor Lattice", The American Nuclear Society Annual Meeting, June 2008, Anaheim (2008).
13. G.D. Del Cul, L.D. Trowbridge, J.P. Renier, R.J. Ellis, K.A. Williams, B.B. Spencer, E.D. Collins, "Analysis of the Reuse of Uranium Recovered from the Reprocessing of Commercial LWR Spent Fuel", ORNL/TM-2007/207, ORNL/GNEP/LTR-2008-002, Oak Ridge National Laboratory (2009).
14. Ronald J. Ellis, "Prospects of Using Reprocessed Uranium in CANDU Reactors, in the US GNEP Program," The American Nuclear Society and the European Nuclear Society 2007 International Conference on Making the Renaissance Real, November 11–15, 2007, Washington, D.C. Trans. Am. Nucl. Soc. 97, 107–108 (2007).
15. J.C. Gehin, J.J. Carbajo, R.J. Ellis, "Issues in the Use of Weapons-Grade MOX Fuel in VVER-1000 Nuclear Reactors: Comparisons of UO₂ and MOX Fuels", ORNL/TM-2004/223, Oak Ridge National Laboratory, October 2004.
16. R.J. Ellis, "Pu Disposition in Russian VVERs: Physics Studies of Lead Test Assembly Designs", paper presented at PHYSOR 2000: ANS International Topical Meeting on Advances in Reactor Physics and Mathematics and Computation, Pittsburgh, PA, May 7-11, 2000.
17. R.J. Ellis, "System Definition Document: Reactor Data Necessary for Modelling Plutonium Disposition in Catawba Nuclear Station Units 1 and 2", ORNL/TM-1999/255, Oak Ridge National Laboratory (2000).